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R I D S / A D D S

SUBJECT: LER 94-009-00:on 940419,CI provisions for two containment monitoring systems were inoperable due to inboard valves not functioning as CI valves in event of LOCA.Corrective action:replace EFC valves w/check valves.W/940519 ltr.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

P.O. Box 968 • 3000 George Washington Way • Richland, Washington 99352

May 19, 1994
GO2-94-121

Docket No. 50-397

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: **NUCLEAR PLANT WNP-2, OPERATING LICENSE NPF-21**
LICENSEE EVENT REPORT NO. 94-009-00

Transmitted herewith is Licensee Event Report No. 94-009-00 for the WNP-2 Plant. This report is submitted in response to the reporting requirements of 10CFR50.73 and discusses the items of reportability, corrective action taken, and action taken to preclude recurrence.

Should you have any questions or desire additional information, please call me or H.E. Kook at (509) 377-4278.

Sincerely,

J.V. Parrish

J.V. Parrish (Mail Drop 1023)
Assistant Managing Director, Operations

JVP/CJF/my
Enclosure

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LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Washington Nuclear Plant - Unit 2	0 5 0 0 0 3 9 7	1 OF 5

TITLE (4)

INCORRECT ISOLATION VALVE COMPONENT SELECTION

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBERS(S)
0 4	1 9	9 4	9 4	-- 0 0 9	-- 0 0	0 5	1 8	9 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

POWER LEVEL (10): 6 3

20.402(b)	20.405(C)	50.73(a)(2)(iv)	77.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.73(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
C. J. Foley, Licensing Engineer	AREA CODE 5 0 9 3 7 7 - 4 3 2 5

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
N/A									

SUPPLEMENTAL REPORT EXPECTED (14)	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE) X NO				

ABSTRACT (16)

On 19 April, 1994, it was determined that containment isolation provisions for two containment monitoring systems were inoperable because the inboard valves for the sampling return lines could not function as containment isolation valves in the event of a LOCA. The immediate corrective action taken was to close the outboard containment isolation valves for these two lines. The root cause is that excess flow check (EFC) valves were installed in these two locations during initial plant construction, and the design of the valves is such that they cannot satisfy the containment isolation function required by General Design Criterion (GDC) 56. A plant modification will be executed during the current refueling outage to replace the two EFC valves with simple check valves, which will satisfy GDC 56.

Plant Conditions

Power Level - 63%
Plant Mode - 1 (Power Operation)

Event Description

On April 19, 1994, a licensing engineer, as part of a broad based review initiated as a result of an NOV related to testing of excess flow check valves, determined that the as-installed configuration of the inboard isolation valves for two return sampling lines penetrating primary containment would not satisfy the requirements for containment isolation, as outlined in General Design Criterion (GDC) 56, "Primary Containment Isolation." These sampling lines are used for containment atmospheric monitoring for reactor coolant leak detection, and are considered process rather than instrument lines.

Immediate Corrective Action

At 1209 hours on April 19, 1994 both inboard isolation valves were declared inoperable. The corresponding outboard isolation valves were closed by deenergizing the circuits that power the solenoid actuators of the outboard valves. The leak detection function was restored to operable status by connecting the sample return line from one monitoring system to the sample supply line to the other monitoring system.

Further Evaluation

EFC valves PI-EFC-X72f and PI-EFC-X73e, used as containment isolation valves, have been inoperable since the time of initial plant startup because the containment leakage limit could have been exceeded in the event of a LOCA coincident with failure of the corresponding outboard containment isolation valve.

Two containment air monitoring systems are installed at WNP-2 as reactor coolant pressure boundary leak detection systems. The systems are installed in cabinets located in the secondary containment area, and draw samples of primary containment atmosphere through radiation monitors, returning the samples to the primary containment. The monitoring systems outboard of the outer containment isolation valves are not qualified as extensions of containment, but are equipped with pressure switches set to isolate the supply and return lines from the monitoring system if the internal pressure in the monitoring system should rise to approximately 2 psig.

In 1981, containment isolation features were subjected to a design review by the Supply System. In the case of these two monitoring systems, it was found that a simple check valve should be installed inside primary containment in the sample return lines to make the installation conform to GDC 56. Directions were given to the architect-engineer to include such check valves in the sample lines; this direction was passed on to the contractor responsible for detailed design and installation of plant instrumentation by way of approved design changes. EFC valves were installed in lieu of simple check valves, but no documentation has been found to explain the bases for that substitution.

Each sample return line is equipped with a Dragon Model 12583A EFC valve installed inside containment, and a spring-loaded solenoid isolation gate valve installed outside of containment. The solenoid valve is automatically closed by the Nuclear Steam Supply Shutoff (NSSS) System in event of a high drywell pressure or low-low reactor water condition. Drywell pressure in excess of approximately 15 psig could actuate the EFC valve, since the valve is designed to close with a through-flow of 4 scfm at 15 psid. However, differential pressure/flow conditions adequate to close the EFC valve could only be established by failure of the corresponding outboard isolation valve, coupled with leakage from the sampling system to secondary containment. In that event, the valve poppet would move to seat against a sealing surface and terminate flow. However, the valve poppet is spring-loaded such that it would automatically move away from the seat and reopen when the differential pressure drops to approximately 3.5 psid. Consequently, the current configuration of the sampling return lines does not meet containment isolation requirements.

A review was performed to determine if other such non-conforming configurations exist at WNP-2. Technical Specification Table 3.6.3-1 identifies all containment isolation valves. The characteristics of each EFC valve were reviewed by comparison with the requirements outlined in FSAR Section 6.2.4.3.2.2 "Evaluation Against Criterion 56" and FSAR Section 6.2.4.3.2.4 "Evaluation Against Regulatory Guide 1.11, 1971, Rev. 0," and the data presented in both FSAR Table 6.2-16 and the Master Equipment List. Applicable surveillance procedures were checked to ascertain how these valves were being tested. Based on this review, it was concluded that the containment monitoring system sampling return lines are the only application of EFC valves required to and failing to satisfy GDC 56 criteria. The remaining EFC valves are installed in instrument lines subject to Regulatory Guide 1.11 criteria, and are not intended to meet GDC 56 criteria.

Root Cause

The root cause is that Dragon Model 12583A excess flow check (EFC) valves were installed in these two locations during initial plant construction, and the design of the valves is such that they cannot satisfy the containment isolation function required by General Design Criterion (GDC) 56.

Further Corrective Actions

The configuration of the sampling return lines will be modified by a design change during the current outage. The modification will replace the EFC valves with simple check valves.

Safety Significance

The monitoring systems encompassing the EFC valves are designed to provide information to control room operators to allow them to take appropriate action in event of reactor coolant pressure boundary leakage. The monitoring systems are not themselves safety-related, and have no control function. The only automatic safety function is that of containment isolation, as initiated by the NSSS system.

Leakage through the containment isolation valves would pass through stainless steel tubing to the sampling/monitoring systems, which would be automatically isolated by internal pressure switches set at approximately 2 psig. Leakage from the tubing or from the monitoring system itself would enter the secondary containment. Thus, a number of barriers exist to guard against the postulated uncontrolled release of radioactivity in the event the EFC valves were to not close:

- The first barrier beyond the EFC valve is the solenoid-operated containment isolation valve. It would be closed by the NSSS system upon detection of a LOCA. These valves are subjected to periodic surveillance tests to confirm operability and leak-tightness, designed to withstand seismic events, and are qualified to withstand both direct and environmental conditions resulting from postulated accidents.

- The second barrier is the radiation sampling/monitoring system. Leakage through the outer containment isolation valve would enter this closed loop system. While it is not qualified as an extension of containment, it would act to limit leakage postulated through the containment isolation valves.

- The third barrier is the secondary containment. Leakage from the solenoid actuated containment isolation valves, the sampling tubing, or from the sampling/monitoring systems would be controlled by the design features used to provide secondary containment.

An analysis was performed using a commercially available code to estimate the effects of a postulated radioactive release, assuming a small break LOCA coincident with:

- failure of both EFC valves to close, and

- failure of one outboard solenoid containment isolation valve to close, and

- breach of the "closed loop" containment monitoring system outboard of the failed-open containment isolation valves, allowing primary containment atmosphere to enter the secondary containment.

Pursuant to Regulatory Guide 1.3 guidelines, the analysis assumes secondary containment protection starting ten minutes after the event, but direct release to the environment during the first ten minutes.

These assumptions result in an effective containment leak rate of 2.17%/day (as compared to the 0.5%/day design bases), with consequent site boundary thyroid and a control room thyroid doses in excess of GDC 19 and 10CFR100 limits. However, the analytical assumptions are very conservative, representing a "worst case" scenario. The probability of occurrence of the small break LOCA is 8×10^{-3}/year, and the probability of the spring loaded solenoid containment isolation valve to close is 1.2×10^{-2}/year. (Figures taken from WNP-2 probabilistic risk studies). The aggregate probability is the product of those numbers, approximately 1×10^{-4}/year. Additionally, the analysis includes the assumption that the "closed loop" monitoring system also fails. While that system is not qualified as an extension of containment, it does exist and functions as a pressure boundary during normal plant operations. It would automatically isolate at 2 psig internal pressure thereby limiting the extent of the system exposed to the primary containment atmosphere as a result of the postulated events. The probability of failure of the pressure boundary of this system is low, but because it has not been considered in probabilistic risk studies at WNP-2 the probability is not quantifiable. But some reduction of the aggregate probability of 10^{-4}/year is present. Based on these additional considerations, it is concluded that the inappropriate installation of EFC valves for use as inboard containment isolation valves was of minor safety significance.

Similar Events

No similar cases have been identified. LER 84-010 was written when four EFC valves failed surveillance testing. Investigation revealed that water service EFC valves had been installed in lines intended for gas service. This is also a component misapplication, but in those cases, the EFC valves were installed in RG 1.11 instrument lines, whereas in the instant case the EFC valves were installed in process lines subject to GDC 56 requirements.

EIIS Information

Text Reference	EIIS Reference System	EIIS Reference Component
Primary Containment	NH	
Leak Detection System	IJ	
PI-EFC-X72f	IJ	V
PI-EFC-X73e	IJ	V